EGG-TMI-6859

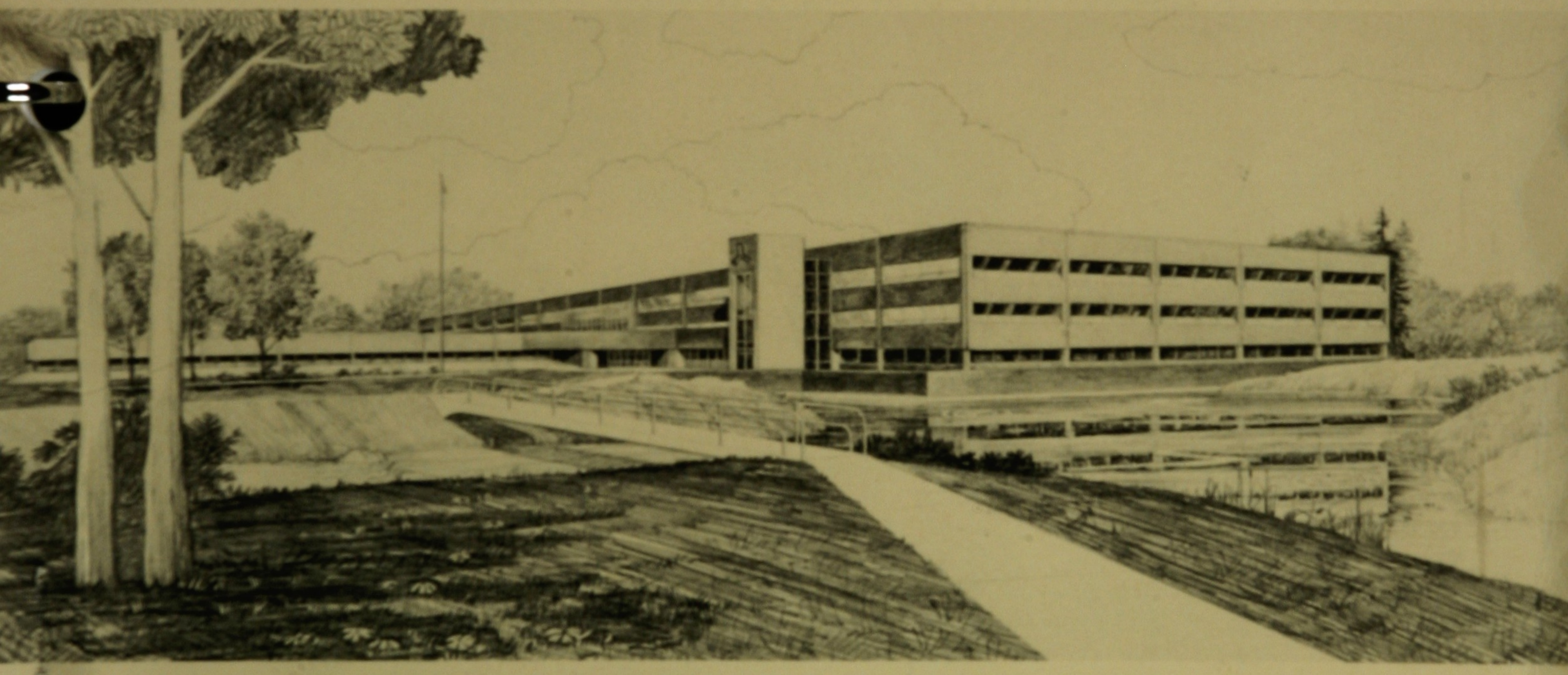
May 1985

COMPUTER CODE CALCULATIONS OF THE TMI-2 ACCIDENT: INITIAL AND BOUNDARY CONDITIONS

Stephen R. Behling

Idaho National Engineering Laboratory
Operated by the U.S. Department of Energy

Informal Report



Prepared for the
U. S. Department of Energy
Idaho Operations Office
Under DOE Contract No. DE-AC07-76ID01570

EG&G Idaho

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ABSTRACT

Initial and boundary conditions during the Three Mile Island Unit 2 (TMI-2) accident are described and detailed. A brief description of the TMI-2 plant configuration is given. Important contributions to the progression of the accident in the reactor coolant system are discussed. Sufficient information is provided to allow calculation of the TMI-2 accident with computer codes.

ACKNOWLEDGMENTS

This document is the result of consultations with many knowledgeable persons. In particular I would like to recognize the expertise of M. L. Picklesimer, M. R. Martin, and C. D. Fletcher.

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1. INTRODUCTION

This document is intended to provide a compilation of best estimate initial and boundary conditions for the Three Mile Island Unit 2 (TMI-2) nuclear reactor during the March 28, 1979 accident. This information is provided to assist those who are performing analyses of the accident. This report is limited to those conditions that affect the progression of the accident in the reactor coolant system. The containment system and auxiliary building will be presented in later reports.

The TMI-2 accident sequence is described below in Section 2. A brief description of the plant configuration is given in Section 3 followed by the plant condition at the time of the turbine trip that initiated the events that ultimately resulted in severe damage to the core in Section 4. The accident boundary conditions are described in Section 5. These boundary conditions represent operator actions or automatic system actuations that influenced the course of the accident.

Much work has gone into analyzing the TMI-2 accident to date. Much of the information provided in this document is compiled from four sources. The first source is NUREG-0600,[1] a report written by a U.S. Nuclear Regulatory Commission (NRC) task force. The second source is the report by the NRC Special Inquiry Group (Rogovin report).[2] The third and fourth major sources were NSAC-80-1[3] and NSAC-24[4] that describe analyses and interpretations of the accident by an industry sponsored group. Additional information has been taken directly from data recorded on the plant reactimeter, a recording device that was operating during the TMI-2 accident and from plant drawings.

This document has been produced by the TMI Accident Evaluation Program of EG&G Idaho for the U.S. Department of Energy, Idaho Operations Office. For any additional information or comments, please contact the author or the EG&G Idaho TMI Accident Evaluation Program.

2. THE TMI-2 ACCIDENT

The TMI-2 accident began when the turbines tripped off and the main feedwater to the steam generators was automatically stopped. The reactor primary system began to heat up and the pressure rapidly increased such that the pilot-operated relief valve (PORV) on the top of the pressurizer opened, the reactor scrammed, and the pressure decreased. This valve failed open at this time, but the operators did not realize it.

The auxiliary feedwater that normally would begin injecting into the steam generator, could not, because two auxilary feedwater block valves were improperly closed. Without auxiliary feedwater the steam generators dried out. This loss of heat sink caused the primary system fluid to continue to heat up, and as this fluid expanded, the pressurizer filled with liquid. This caused, as the pressurizer continued to read above normal, the operators to believe the primary system was full and for the next few hours they defeated the injection systems that could have replaced the water being lost out the PORV.

Eventually, sufficient water left the primary system to cause the reactor coolant pumps, that were now pumping a mixture of steam and liquid water, to alarm on high vibrations. The pumps were turned off, the last ones at 1 h 40 min after turbine trip, and the mixture of steam and liquid separated, resulting in a reactor vessel liquid level near the top of the core. The liquid level continued to decrease as the boil-off rate was greater than the fluid injection rate (the pressurizer was still indicating higher than normal levels and safety injection was inhibited).

As the level decreased, the decay heat being generated in the fuel was not removed and the uncovered portions of the fuel rods began to heat up. The leak through the PORV was discovered and stopped at 2 h 22 min but it was too late to prevent core damage. The heatup in the presence of steam caused an exothermic reaction as the zircaloy fuel rod cladding oxidizes, further heating the rods and producing hydrogen gas. The resulting higher temperatures led to fuel liquefaction (molten zircaloy dissolving uranium dioxide) and fuel melting in some core locations. Sufficient hydrogen was produced by oxidation that a hydrogen burn occurred in the containment almost 10 h after the turbine trip.

Post-accident examination has determined that during the accident sufficient fuel damage occurred to result in a region void of fuel that encompasses about one-third of the core volume. Fuel liquefaction and relocation resulted in a large quantity of once molten material to settle in the vessel lower plenum.

The core and relocated core materials were eventually cooled by 16 h after the turbine trip. Many actions took place during those 16 h that changed or exacerbated the accident. Those events are described in Section 4 of this document.

3. SUMMARY OF PLANT CONFIGURATION

The TMI-2 plant configuration is presented in this section. These descriptions are intended to orient the reader with most of the components that were important to the progression of the accident in the primary system. For more detailed descriptions of the plant the reader is referred to References 2 and 3 and the plant final safety analysis report.

The layout of the plant can be seen in Figure 1. The reactor vessel is connected to two loops, the A loop and the B loop, each having a hot leg and two cold legs. There are two once-through steam generators, one per loop, and a pressurizer connected to the A loop.

3.1 Configuration of the Reactor Vessel

The reactor vessel contains the core. The water from the cold legs is directed through the downcomer and lower plenum, past the fuel elements in the core, into the upper plenum, and out the hot legs. Approximately 10.4% of the total flow entering the reactor vessel bypasses the fuel regions of the core during normal operation. A flow path through the control rod guides and instrument tubes allows 6.9% of the total flow to bypass direct contact with the fuel elements, however, this fluid is heated by the radiation fields in this area. Approximately 1.5% of total flow flows between the core former plates and the core barrel region. About 2% flows through leakage paths around the hot leg nozzles between the downcomer and the upper plenum.

A set of vent valves exists between the upper plenum and downcomer. These valves automatically open under a positive pressure gradient from upper plenum to downcomer. The purpose of the vent valves is to prevent steam binding in the upper plenum and the resultant core liquid level depression during the reflood phase of a large loss-of-coolant accident. A differential pressure of 0.1 psi is needed to begin opening the valves and a differential pressure of 0.25 psi can completely open the valves.

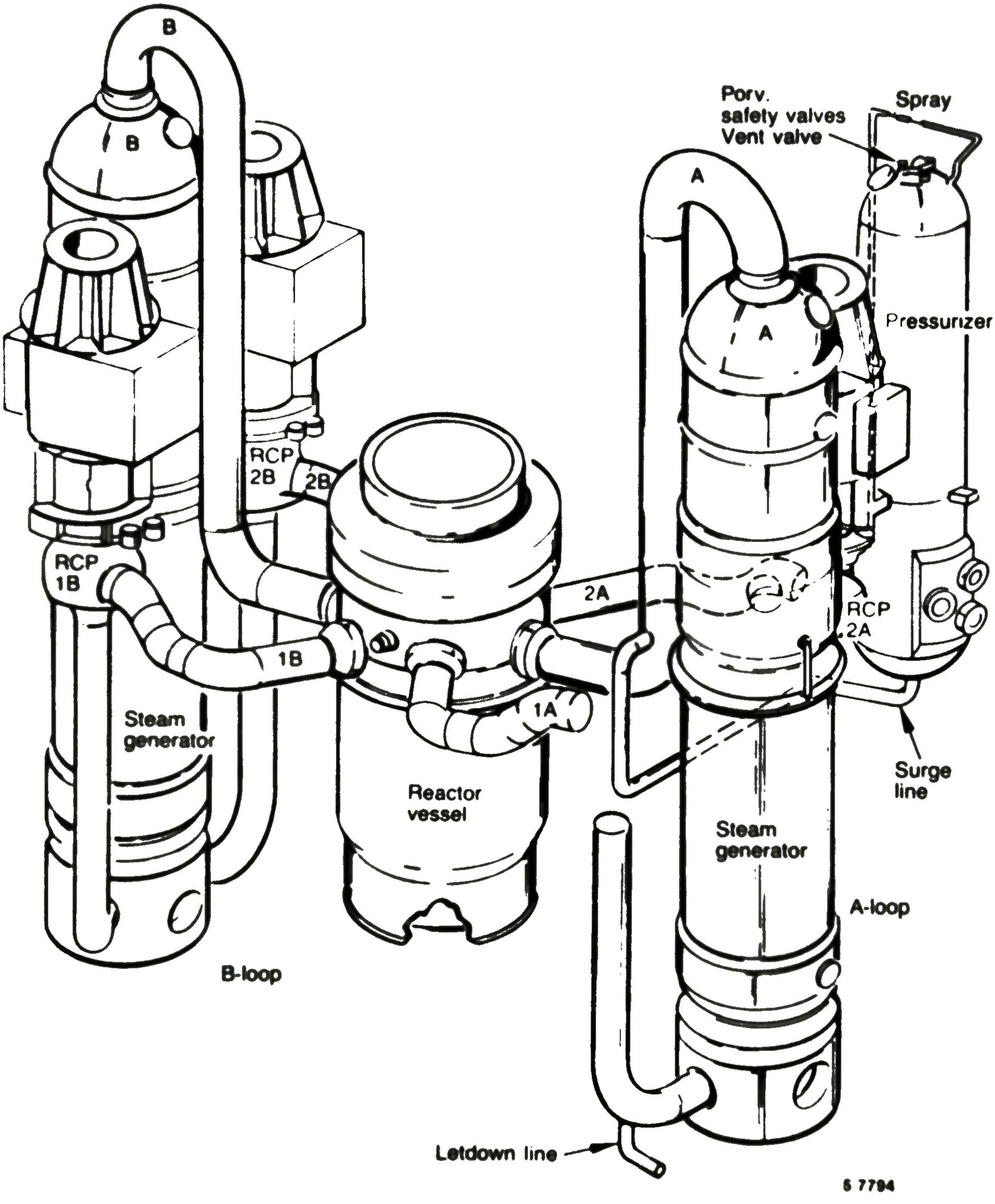

Figure 1. Schematic drawing of the TMI-2 reactor coolant system.

The core flood tanks inject into the upper portion of the downcomer. During the TMI-2 accident, the flood tanks injected only a small amount of water more than 8 h after the turbine trip that initiated events.

3.2 Configuration of the Hot Legs

The two hot legs connect the reactor vessel to the top of the steam generators. The upper portion of the hot leg is often referred to as the candy cane. The two hot legs are referred to as the A loop and B loop, and the pressurizer is connected to bottom of the candy cane on the A loop. The primary system pressure is measured near the top of the B loop hot leg. The temperature of each hot leg is also measured near the top of the candy cane. The loop flows are measured with venturi flow meters in the candy cane.

The orientation of the pressurizer surge line connection to the side of the A loop hot leg can be seen in Figure 1. The connection is just downstream of a 90-degree bend. The phase separation of the relatively high velocity liquid-vapor stream in this bend during two-phase pump operation should be accounted for when calculating the flow into the surge line.

3.3 Configuration of the Pressurizer

The pressurizer, Figure 2, controls the reactor coolant system pressure during normal operation. The pressure is decreased by turning off the pressurizer heaters and by opening the spray valve allowing water from the cold leg to spray into the steam space and condense some vapor. The spray operates on pressure differential between the outlet of reactor coolant pump 2A and the pressurizer. Therefore, the spray operates normally only when pump 2A is running. Pressure is increased by energizing the pressurizer heaters causing some boiling of pressurizer liquid.

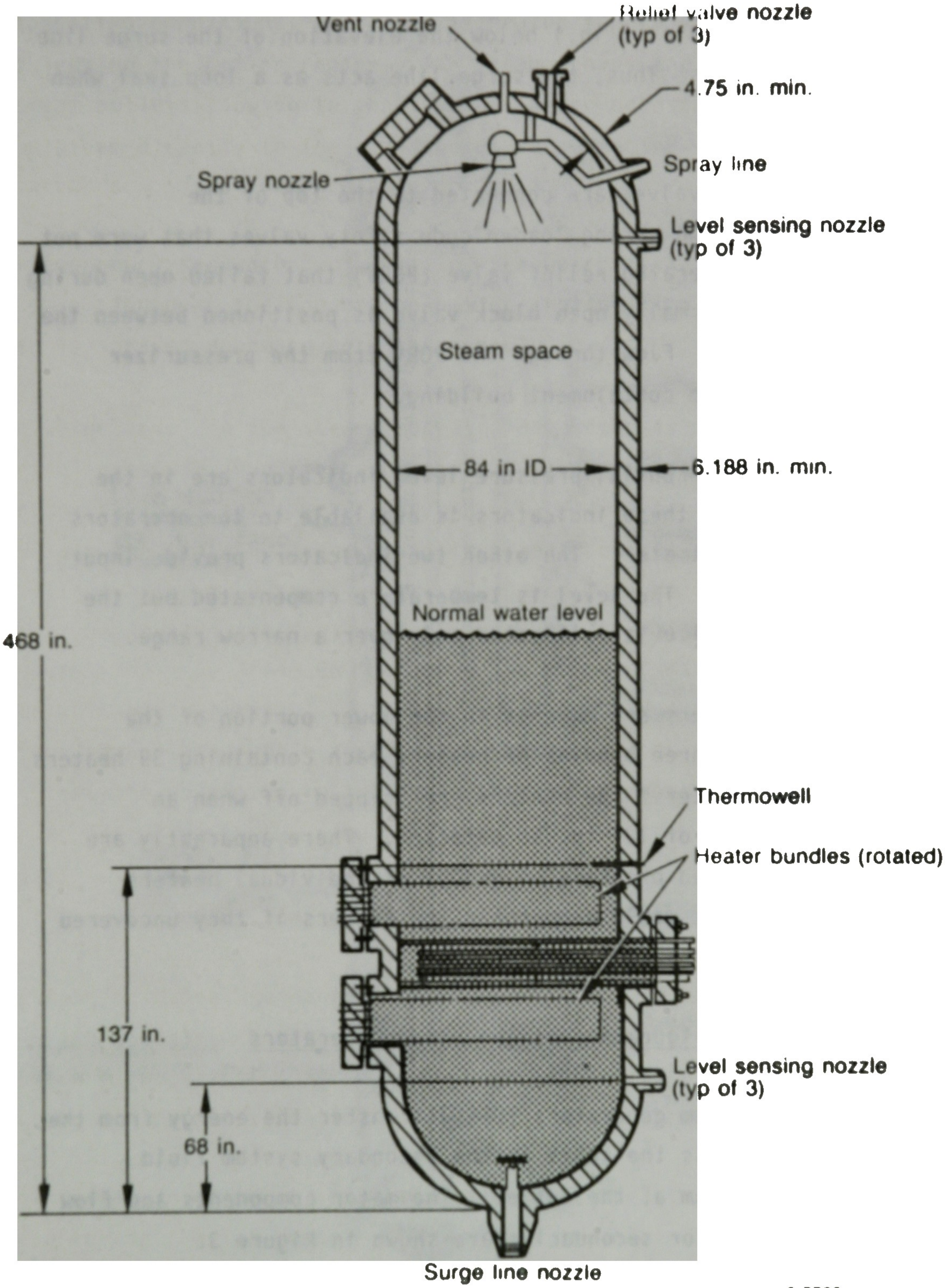

Figure 2. TMI-2 pressurizer.

The pressurizer is connected to the A-loop hot leg by the pressurizer surge line, entering the bottom of the pressurizer through the surge line nozzle. The reference elevation of the bottom of the pressurizer is 310 ft. This is 3.5 m (11 ft 6 in.) below the elevation of the surge line connection to the hot leg. Thus, the surge line acts as a loop seal when liquid filled.

The pressure relief valves are connected to the top of the pressurizer. There are two spring loaded code safety valves that were not challenged and a pilot-operated relief valve (PORV) that failed open during the TMI-2 accident. A normally open block valve is positioned between the pressurizer and the PORV. Flow through the PORV from the pressurizer enters a drain tank in the containment building.

Three identical differential pressure level indicators are in the pressurizer. Only one of these indicators is available to the operators and recorded on the reactimeter. The other two indicators provide input for plant safety systems. The level is temperature compensated but the temperature measuring device is calibrated only over a narrow range.

The pressurizer heaters are mounted in the lower portion of the pressurizer. There are three bundles of heaters each containing 39 heaters that supply 14 kW per heater. The heaters are tripped off when an indicated low water level of 127 in. is detected. There apparently are thermostatically controlled breakers on groups of individual heaters (9 heaters per group) that would deenergize the heaters if they uncovered and overheated.

3.4 Configuration of the Steam Generators

The once-through steam generators (OTSG) transfer the energy from the primary system fluid across the tubes to the secondary system fluid producing superheated steam at the outlet. The major components and flow paths in the steam generator secondaries are shown in Figure 3.

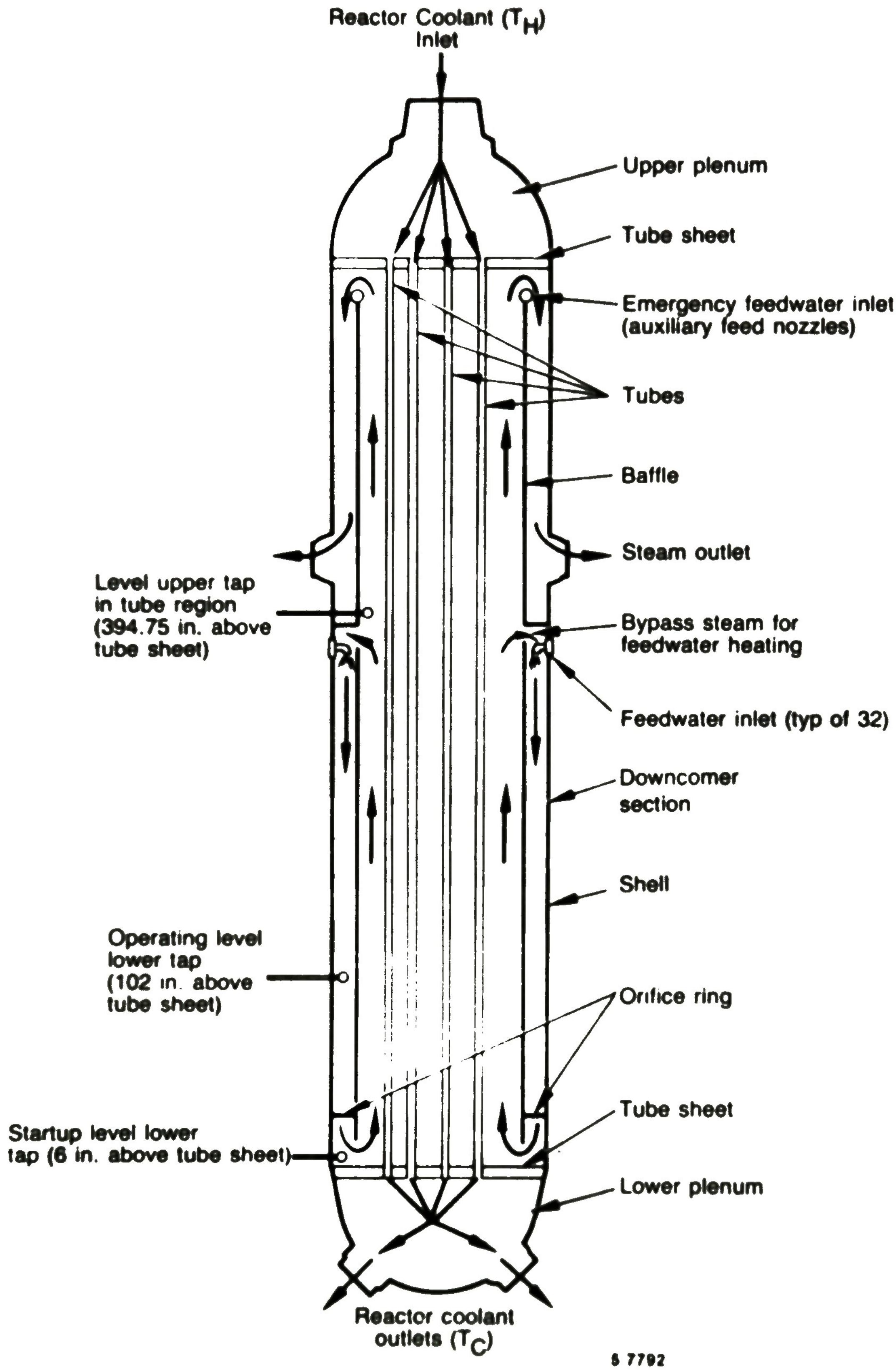

Figure 3. TMI-2 once through steam generator.

The feedwater enters the steam generator through 32 nozzles and is mixed with fluid from the boiler region. This fluid flows down a downcomer, through an orifice plate, and enters the boiler region. At normal operating conditions, the fluid is heated to dry superheated steam prior to leaving the boiler region. The steam leaves the steam generator at the steam outlets flowing to the turbines and condenser. Relief valves that dump steam directly to the atmosphere are connected to this line to prevent secondary system overpressure.

The emergency feedwater or auxiliary feedwater is injected through nozzles on a ring near the top of the boiler region. The auxiliary feedwater is sprayed downward onto the tubes.

The water level in the steam generator downcomer is measured using differential pressure level transducers. Two measurements that were recorded on the TMI-2 reactimeter during the accident for each steam generator are the start-up range level that measures the downcomer level from 6 in. above the tube sheet in the downcomer to a height 394.75 in. above the tube sheet in the boiler region and the operating range level that measures the level from 96 in. above the start-up range lower tap to the start-up range upper tap. The operating range level measurement is temperature compensated while the startup range level measurement is not. The fact that the upper tap is in the boiler region means the instrument readings are effected by the flow friction.

3.5 Configuration of the Cold Legs

The four cold legs connect the steam generators to the reactor vessel. Each cold leg contains a reactor coolant pump (RCP) and an emergency core coolant (ECC) injection port. Both high pressure injection (HPI) during off-normal situations and makeup flow during normal or off-normal situations are injected through these ports. Usually, makeup flow is injected into the 1B cold leg. Note that the elevation of the connection to the reactor vessel is lower than the elevation of the reactor coolant pumps. The letdown line is connected to the bottom of the 1A cold leg. The 2A cold leg contains the pressurizer spray line connection, connected to the pump casing at the pump exit. Cold leg temperatures are measured about 3 ft upstream of the pump inlets.

The RCPs are not self-priming. That is, the pumps cannot develop a pressure head unless there is water in the impeller region.

4. TMI-2 PLANT CONDITIONS AT TURBINE TRIP

The conditions of the TMI-2 plant at the time of the turbine trip that initiated the accident are given in Table 1. These values have been taken from the reactimeter and other sources. At the time of the turbine trip the pressurizer was being operated in manual mode. All other major systems were in automatic mode.

TABLE 1. TMI-2 PLANT INITIAL CONDITIONS

Reactor power	97.2%, (2694 MW)
Hot leg temperature	592 K (606°F)
Cold leg temperature	565 K (557°F)
Reactor coolant flow rate	1.71×10^4 kg/s (3.81×10^4 lbm/s)
Pressurizer level	5.8 m (229 in.)
Pressure (B-loop hot leg)	15.91 MPa (2307 psia)
A-loop steam generator	
Pressure	6.36 MPa (923 psia)
Level (above tube sheet)	6.76 m (266 in.) operating level instrument
Level (above tube sheet)	4.22 m (166 in.) startup level instrument
B-loop steam generator	
Pressure	6.34 MPa (920 psia)
Level (above tube sheet)	6.85 m (270 in.) operating level instrument
Level (above tube sheet)	4.29 m (169 in.) startup level instrument
Steam generator feedwater temperature	513 K (463°F)

5. TMI-2 ACCIDENT BOUNDARY CONDITIONS

The boundary conditions that affected the progression of the TMI-2 accident in the reactor coolant system are presented in this section. The net mass loss from the primary system, events in the pressurizer, the reactor coolant pump behavior, and steam generator conditions are described.

5.1 Net Mass Loss from Primary System

During the TMI-2 accident, sufficient water was lost from the primary system to cause the core damage. The system water mass losses during the accident were the flow out of the pressurizer PORV, the letdown flow, and system leakage. The system water mass gains during the accident were the makeup and high pressure injection (HPI) flows, the pump seal injection, and the core flood tank. Each of these losses or additions is described in more detail below.

The core began to heat up shortly after the A-loop primary coolant pumps were tripped off 101 min after the turbine trip based on measured hot leg temperatures. The total net mass lost out of the system prior to pump trip had to be sufficient to result in a water level near or below the top of the core following pump trip as the vapor and liquid separated. If one makes assumptions about the amount of liquid that settles into the pump suction portion of the cold legs following pump trip, one can calculate how much mass had to have left the system. In order to ensure that a calculation of the early phases of the accident is within assumed reactor vessel liquid level ranges, the various mass inflows or outflows can be adjusted, thus decreasing the importance of the sometimes large uncertainties in the individual flows.

5.1.1 PORV Flow Rate

The flow out the PORV was a function of the valve position (open/closed), the upstream block valve position, and the fluid condition upstream of the valve. The pressure in the pressurizer was much greater than the sink pressure (drain tank) throughout the accident and the flow through the PORV was critical or choked whenever both the PORV and block valve were open. It should be assumed that the PORV failed open on its first challenge when the pressure exceeded 15.65 MPa (2270 psia) and remained open throughout the accident. A similar valve has been flow tested[2] and the flow is 17.3 kg/s (38.1 lbm/s) for pure steam at 16.24 MPa (2355 psia) and 80.8 kg/s (178.1 lbm/s) for pure liquid at 16.27 MPa (2360 psia). Earlier sources[3] report an area of a 1-5/32 in. orifice (6.774×10^{-4} m^2 or 0.007292 ft^2).

The position of the block valve upstream of the PORV is either open or closed. The position of the block valve as a function of time from turbine trip during the accident is given in Table 2 and has been compiled from Reference 3.

5.1.2 Letdown Flow

The letdown system removes primary system fluid from the bottom of the pump suction of the 1A cold leg. Letdown flow removed significant amounts of fluid from the system during the TMI-2 accident.

Two differing sources of letdown flow history information are the operator testimony following the accident and measured letdown cooler behavior. The operators stated that throughout the accident they tried to maintain letdown at about 60 gal per minute (measured downstream of the coolers). [3.8 kg/s (8.4 lbm/s)].

The letdown flow has also been estimated using the temperature response of the two letdown coolers and the primary system temperature. The sources for the letdown flow are NSAC-80-1[3] in which the flow is given from 0 to 7000 s, and NSAC-24[4] in which the flow is given from 6000 to 12000 s. Unfortunately, the calculated flow during the common time period is different in the two sources. The NSAC-80-1 flow information is contained in a plot (in units of lbm/h) for the results of a single letdown cooler. The flow from the figure must be doubled to account for both letdown coolers (assuming identical response). The resulting flow at

TABLE 2. PRESSURIZER BLOCK VALVE OPERATION

Time (h:min after turbine trip)	Time (s after turbine trip)	Valve Operation
0	0	Open
2:22	8520	Closed
3:12	11520	Open
3:17	11820	Closed
3:40	13200	Open
5:18	19080	Closed
5:40 to 7:38	20400 to 27480	Cycled open and closed to maintain pressure between 13.2 and 14.9 MPa (1915 and 2165 psia)
7:38	27480	Open
~9:10	~33000	Closed
~9:25	~33900	Open
9:49	35340	Closed
10:00	36000	Open
11:08	40080	Closed
12:36	45360	Open
12:47	46020	Closed
12:52	46320	Open
13:00	46800	Closed

6000 s from NSAC-80-1 is 10.6 kg/s (23.3 lbm/s). The NSAC-24 flow information is contained in a figure (in units of kg/s) for total letdown flow. The resulting flow at 6000 s from NSAC-24 is 7.30 kg/s (16.1 lbm/s). NSAC-24 refers to potential fouling of the letdown coolers (prior to the accident) as an adjusting factor for calculating the flow. It is felt that the NSAC-24 reported flow is more accurate.

Based on the above information, the flow from NSAC-80-1 from 0 to 6000 s is modified to be consistent with NSAC-24 by dividing it by 1.45. NSAC-24 flow is used as reported from 6000 to 12000 s. The resultant flow time history is listed in Table 3. The average flow rate to 12000 s from Table 3 is 5.67 (12.5 lbm/s).

5.1.3 System Leakage

The system leakage had been determined to be 6 gpm liquid equivalent prior to the accident. Because elevated temperatures were observed downstream of the PORV, it is felt most of this leakage occurred there. When the PORV failed open this leakage can be ignored. Any additional leakage can also be ignored as it would have been well within the uncertainties of other mass losses and additions. There is no evidence that any additional leaks developed during the course of the accident.

5.1.4 Makeup and HPI Flows

The makeup and HPI flows during the TMI-2 accident have a large uncertainty. During makeup the flow can be, and was, manually throttled allowing flow to bypass the injection port and flow back into the makeup storage tank. The most probable estimates for flows have a large uncertainty such that the difference between a higher estimated flow and a lower estimated flow over the course of the accident (16 h) is approximately equal to the original water mass in the reactor primary coolant system. It is probable that any detailed calculations of the TMI-2 accident will require two analyses, with different injection flow assumptions, in order to bound the possible behaviors.

TABLE 3. TMI-2 LETDOWN FLOW HISTORY BASED ON LETDOWN COOLERS

Time (s)	Flow Rate (kg/s)	Flow Rate (lbm/s)
0	1.91	4.21
300 (-)	1.91	4.21
300 (+)	8.17	18.01
550 (-)	8.17	18.01
550 (+)	1.91	4.21
900 (-)	1.91	4.21
900 (+)	8.17	18.01
1400 (-)	8.17	18.01
1400 (+)	1.91	4.21
1700 (-)	1.91	4.21
1700 (+)	7.30	16.09
2400 (-)	7.30	16.09
2400 (+)	1.91	4.21
2800 (-)	1.91	4.21
2800 (+)	8.34	18.39
4600 (-)	8.34	18.39
4600 (+)	1.91	4.21
4950 (-)	1.91	4.21
4950 (+)	6.60	14.56
5400 (-)	6.60	14.56
5400 (+)	1.91	4.21
5650 (-)	1.91	4.21
5650 (+)	7.30	16.09
7500 (-)	7.30	16.09
7500 (+)	2.00	4.41
7800 (-)	2.00	4.41
7800 (+)	0.0	0.0
8700 (-)	0.0	0.0
8700 (+)	2.00	4.41
9600 (-)	2.00	4.41
9600 (+)	8.00	17.64
12000 (-)	8.00	17.64
12000 (+)	0.0	0.0

The operators' testimony indicates that during the accident the injection flow was throttled to a level that just replaced letdown and assumed leakage. This flow is about 66 gpm--4.2 kg/s (9.2 lbm/s). Other information provides the makeup/HPI pump operation history.

The makeup/HPI system consists of three pumps and many lines and valves. When in makeup mode, the injection is normally lined up to the 1B cold leg. When one pump is running in makeup mode the most likely injected flow during the accident is felt to be between a throttled flow of 66 gpm and a full flow of 160 gpm [4.17 and 10.04 kg/s (9.19 and 22.14 lbm/s)]. When two pumps are running in makeup mode, the most likely injected flow is felt to be between 250 and 300 gpm [15.69 and 18.83 kg/s (34.60 and 41.52 lbm/s)]. These values assume that the operators did not manually throttle the flow to 66 gpm when they manually started a second makeup pump.

When in HPI or engineered safeguards (ES) mode, two pumps are running and valves are set such that 1000 gpm [62.77 kg/s (138.4 lbm/s)] are injected with one-quarter of the total flow entering each of the four cold leg injection ports. During the first 16 h of the accident the total time the ES mode was on amounts to less than 15 min. Some operator testimony implies that even when in ES mode, the flow was throttled shortly after initiation.

The operation history of the makeup pumps has been taken from NUREG-0600,[1] in which actions were determined from the alarm printers, operating logs, and interviews. The resultant injection flows using pump operation history and operator testimony as a function of time after turbine trip are given in Table 4.

Changes in the level of the borated water storage tank have been used to estimate injection flows and these flows are generally much larger than shown in Table 4. However, since a large amount of the fluid can bypass the reactor coolant system and return to the makeup tank when ES mode is defeated, it is not felt to be a reliable indication of injected flow.

TABLE 4. MAKEUP AND HPI INJECTION RATES DURING THE TMI-2 ACCIDENT

Time after Turbine Trip		Injection Flow[a]		
(h:min:s)	(s)	(gpm)	(kg/s)[b]	(lbm/s)[b]
0	0	40	2.51	5.54
00:00:41	41	250-300	15.69-18.83	34.60-41.52
00:02:02	122	1000[c]	62.77[c]	138.4[c]
00:04:38	278	66-160	4.17-10.04	9.19-22.14
00:10:24	624	0.0	0.0	0.0
00:11:43	703	66-160	4.17-10.04	9.19-22.14
03:20:00	12000	1000[c]	62.77[c]	138.4[c]
03:37	13020	66-160	4.17-10.04	9.19-22.14
03:56	14160	1000[c]	62.77[c]	138.4[c]
04:00	14400	250-300	15.69-18.83	34.60-41.52
04:17	15420	0.0	0.0	0.0
04:22	15720	66-160	4.17-10.04	9.19-22.14
04:27	16020	250-300	15.69-18.83	34.60-41.52
09:04	32640	66-160	4.17-10.04	9.19-22.14
09:50	35400	1000[c]	62.77[c]	138.4[c]
09:51	35460	66-160	4.17-10.04	9.19-22.14
10:32	37920	250-300	15.69-18.83	34.60-41.52
10:36	38160	66-160	4.17-10.04	9.19-22.14
11:19	40740	250-300	15.69-18.83	34.60-41.52
11:28	41280	66-160	4.17-10.04	9.19-22.14
11:33	41580	250-300	15.69-18.83	34.60-41.52
11:36	41760	66-160	4.17-10.04	9.19-22.14
13:23	48180	250-300	15.69-18.83	34.60-41.52
14:41	52860	270[d]	16.95	37.36
14:43	52980	66-160	4.17-10.04	9.19-22.14
15:33	55980	250-300	15.69-18.83	34.60-41.52
15:39	56340	66-160	4.17-10.04	9.19-22.14
15:49	56940	250-300	15.69-18.83	34.60-41.52
15:56	57360	66-160	4.17-10.04	9.19-22.14

a. Injection during makeup is into the 1B cold leg injection port.

b. Assumes 100°F water.

c. HPI mode--injection is equally divided into the four cold leg injection ports.

d. Throttled flow (from NUREG-0600).

5.1.5 Pump Seal Injection

The reactor coolant pump (RCP) seals are maintained and cooled using injected water from the makeup system. Whenever at least one makeup pump is running, the seal flow into the primary system is between 8 and 10 gpm per reactor coolant pump when the RCPs are running. All makeup pumps were off from 10 min 24 s to 11 min 43 s and again from 4 h 17 min to 4 h 22 min after turbine trip (see Table 4). Operator testimony indicates a smaller but nonzero flow was maintained when the RCPs were not rotating.

5.1.6 Core Flood Tank Injection

The core flood tanks are components of a passive system that inject water directly into the reactor vessel downcomer whenever the primary system pressure decreases to below 4.24 MPa (615 psia).

During the TMI-2 accident, about 2.83 m^3 (100 ft^3) of liquid could have been injected into the reactor vessel[3] from 8:31 (30660 s) to 9:10 (33000 s) with by far the largest flow occurring early in this period. This flow averages about 0.96 kg/s (2.13 lbm/s) and is much less than the makeup flow at this time.

5.2 Pressurizer Events

The automatic and operator controlled events in the pressurizer during the TMI-2 accident include the control of the block valve, the pressurizer heaters, the pressurizer sprays, and the pressurizer vent valve. The block valve was discussed in Section 5.1. The heater, vent valve, and spray control are described below.

5.2.1 Pressurizer Heater Behavior

The pressurizer heaters are divided into 5 banks that are further divided into 13 groups. Each bank has a low pressure set point for turning the heaters on and a high pressure setpoint for turning the heaters off when in automatic mode. The configuration is detailed in Table 5.

The alarm printer printed actions related to groups (not banks) of heaters during the accident. These actions have been compiled in Table 6. During the course of the accident some heater groups apparently failed while no indication of any action for some groups was printed. For much of the accident, the pressure was low enough such that all the heaters should have been on.

Another hypothesis is that the pressurizer heaters were periodically covering and uncovering and the thermostats (see Section 3.3) were tripping the heaters on and off. This requires that the pressurizer level instrumentation was indicating incorrectly. The instrument error is possible if steam or hydrogen entered the pressurizer level instrument reference leg at the upper level sensing nozzle (see Figure 2), but this possibility has not yet been demonstrated. Should this hypothesis be true, then the heaters would provide an indication of pressurizer liquid level. Based on a plant drawing, heater banks 1 and 2 (groups 12 and 13) are in the lowest elevation heater bundle. These heaters were apparently operational throughout the accident. Heater banks 4 and 5 (groups 1 through 7) are in the upper bundles and these groups tripped off and on throughout the accident.

For performing analysis using heater power, it is recommended that Table 6 and the setpoints from Table 5 be logically combined, except for those heaters that apparently remain off.

5.2.2 Pressurizer Spray Behavior

The pressurizer spray is designed to decrease the primary system pressure by spraying cold water from the outlet of the 2A cold leg reactor coolant pump to the steam space in the pressurizer. The spray valve automatic controller opens the spray valve when the hot leg pressure is greater than 2205 psig (2220 psia or 15.31 MPa). The spray valve should close whenever the pressure decreases to below the setpoint of 2155 psig (2170 psia or 14.96 MPa).

TABLE 5. TMI-2 PRESSURIZER HEATER CONFIGURATION

Heater Bank	Corresponding Heater Group Number(s)	Total kW[a]	Low Pressure[b] On Setpoint in psig (MPa)	High Pressure[b] Off Setpoint in psig (MPa)
1	13	126	2147 (14.904)	2155 (14.959)
2	12	126	2135 (14.821)	2155 (14.959)
3	8, 9, 10, 11	504	2135 (14.821)	2155 (14.959)
4	4, 5, 6, 7	504	2120 (14.718)	2140 (14.856)
5	1, 2, 3	378	2015 (13.994)	2125 (14.752)

a. Each group provides 126 kW.

b. From NSAC-80-1.[3] Pressure is the gauge pressure measured in the A-loop hot leg. Atmospheric pressure is assumed to be 14.7 psia.

TABLE 6. PRESSURIZER HEATER RESPONSE DURING TMI-2 ACCIDENT

Time (h:min:s)	Time (s)	Event
00:00:08	8	All heaters (Groups 1-13) automatic
02:54:19	10459	Heater groups 1, 2, 3, 4, 5 off
04:23:54	15834	Heater groups 1, 2, 3, 4, 5 on
04:30:30	16230	Group 10 off (remains off, assumed failed)
04:46:21	17181	Groups 4, 5 off (remain off, assumed failed)
05:30:34	18034	Group 3 off (remain off, assumed failed)
06:13:39	22419	Groups 1, 2 off
06:14:06	22446	Groups 1, 2 on
07:50:16	28216	Groups 1, 2 off
09:55:10	35710	Group 8 off (remains off, assumed failed)
10:05:25	36325	Groups 1, 2 on
10:07:19	36439	Groups 1, 2 off
10:32:36	37956	Groups 1, 2 on
10:38:57	38337	Groups 1, 2 off
10:39:51	38391	Groups 1, 2 on
11:28:52	41332	Groups 1, 2 off
11:45:17	42317	Groups 1, 2 on
13:26:00	48360	Groups 1, 2 off
14:25:26	51926	Groups 1, 2 on

The position of the valve was recorded on the reactimeter and that data is reproduced in Table 7. The recorded valve cycling from about 20 s to 10 min 21 s is contrary to the setpoints described above. When the pressurizer heaters were placed in automatic at about 8 s, the spray should also have been in automatic. If two calculations are not possible, it is recommended the spray be assumed to have operated in automatic mode prior to 2 h after turbine trip and manually thereafter as listed in Table 7.

The spray is effective only when the 2A pump is running. However, if the valve is open when the pump is off, gases can be transported between the cold leg and the pressurizer bypassing any water seals in the pressurizer surge line and lower reactor vessel.

5.2.3 Pressurizer Vent Valve

The 1-in. diameter vent valve on the top of the pressurizer is normally used only to remove noncondensible gases from the primary system following shut down events such as refueling. According to the Rogovin Report,[2] this valve was cycled three times during the accident. The approximate times after turbine trip were: opened at 7 h 45 min, closed at 9 h 10 min; opened at 10 h 35 min, closed at 11 h 10 min; and opened at 12 h 45 min, closed at 12 h 58 min.

5.3 Reactor Coolant Pump Behavior

The primary coolant pumps were turned off and were restarted at various times during the accident. The events are listed in Table 8. Only events that resulted in fluid being pumped are included. The operators started pumps at other times during the accident, but because they were steam filled, they developed no head, used very little current, and were quickly turned off.

TABLE 7. SPRAY VALVE POSITION FROM REACTIMETER DATA

Time (h:min:s)	Time (s)	Event
00:00:00	0	Open
00:00:12	12	Closed
00:00:39	39	Open
00:00:48	48	Closed
00:01:06	66	Open
00:01:18	78	Closed
00:01:39	99	Open
00:01:48	108	Closed
00:02:12	132	Open
00:02:24	144	Closed
00:02:48	168	Open
00:03:00	180	Closed
00:03:30	210	Open
00:03:45	225	Closed
00:04:06	246	Open
00:04:21	261	Closed
00:04:48	288	Open
00:05:03	303	Closed
00:05:27	327	Open
00:05:42	342	Closed
00:06:06	366	Open
00:06:21	381	Closed
00:07:00	420	Open

TABLE 7. (continued)

Time (h:min:s)	Time (s)	Event
00:07:21	441	Closed
00:07:54	474	Open
00:08:21	501	Closed
00:09:03	543	Open
00:09:18	558	Closed
00:10:03	603	Open
00:10:21	621	Closed
02:55:03	10503	Open
03:13:18	11598	Closed
03:45:21	13521	Open
04:21:42	15702	Closed
07:58:09	28689	Open
09:07:18	32838	Closed
10:04:24	36264	Open
12:05:51	43551	Closed

TABLE 8. PRIMARY COOLANT PUMP OPERATION

Time (h:min:s)	Time (s)	Event
00:00:00	0	All pumps running
01:13:29	4409	1B and 2B pump off
01:40:37	6037	2A pump off
01:40:45	6045	1A pump off
02:54:00	10440	2B pump on
03:12:00	11520	2B pump off
04:08:37	14917	1A pump on
04:09:14	14954	1A pump off
15:32:42	55962	1A pump on
15:32:52	55972	1A pump off
15:49:36	56976	1A pump on

5.4 Steam Generator Conditions

The steam generator conditions changed throughout the TMI-2 accident. The main feedwater flow decreased to zero within 1 s after turbine trip and the steam generators boiled dry by 1 min 45 s. At 8 min and 18 s emergency feedwater began when the mistakenly closed auxiliary feedwater block valves were opened.

The pressure and water level changed throughout the transient. These parameters were recorded on the plant reactimeter and are shown on several figures. Figure 4 shows the pressure in the A-loop steam generator secondary from the start of the accident until 16 h. Figure 5 shows the same information for the first 5 h after turbine trip. Figure 6 and 7 show the pressure in the B-loop steam generator for 0 to 16 h and 0 to 5 h respectively. Figures 8 and 9 are the long term and short term recorded startup level (maximum reading of 250 in.) for the A-loop steam generator. Figures 10 and 11 are the long and short term operating range level for the A-loop steam generator. Figures 12 through 15 are similar figures for the B-loop generator. The operating level reading is temperature compensated while the startup level reading is not. Thus the two level measurements are not identical. It is recommended the operating level be used unless that level is less than about 10% when the startup level should be used (see Section 3.4).

In order to minimize calculational uncertainties, INEL analyses use the measured steam generator pressures and levels as boundary conditions. The flow out of the generator is increased or decreased to match the measured pressure. After 8 min 18 s when an operator opened the improperly closed auxiliary feedwater valves, the auxiliary feedwater flow is increased or decreased to match the measured level. This is done to eliminate the need to estimate the manual adjustments that were made to the auxiliary feedwater flows in each OTSG throughout the accident. The auxiliary feedwater temperature is 311 K (100°F). The maximum auxiliary feedwater flow per steam generator is about 30.8 kg/s (68 lbm/s).

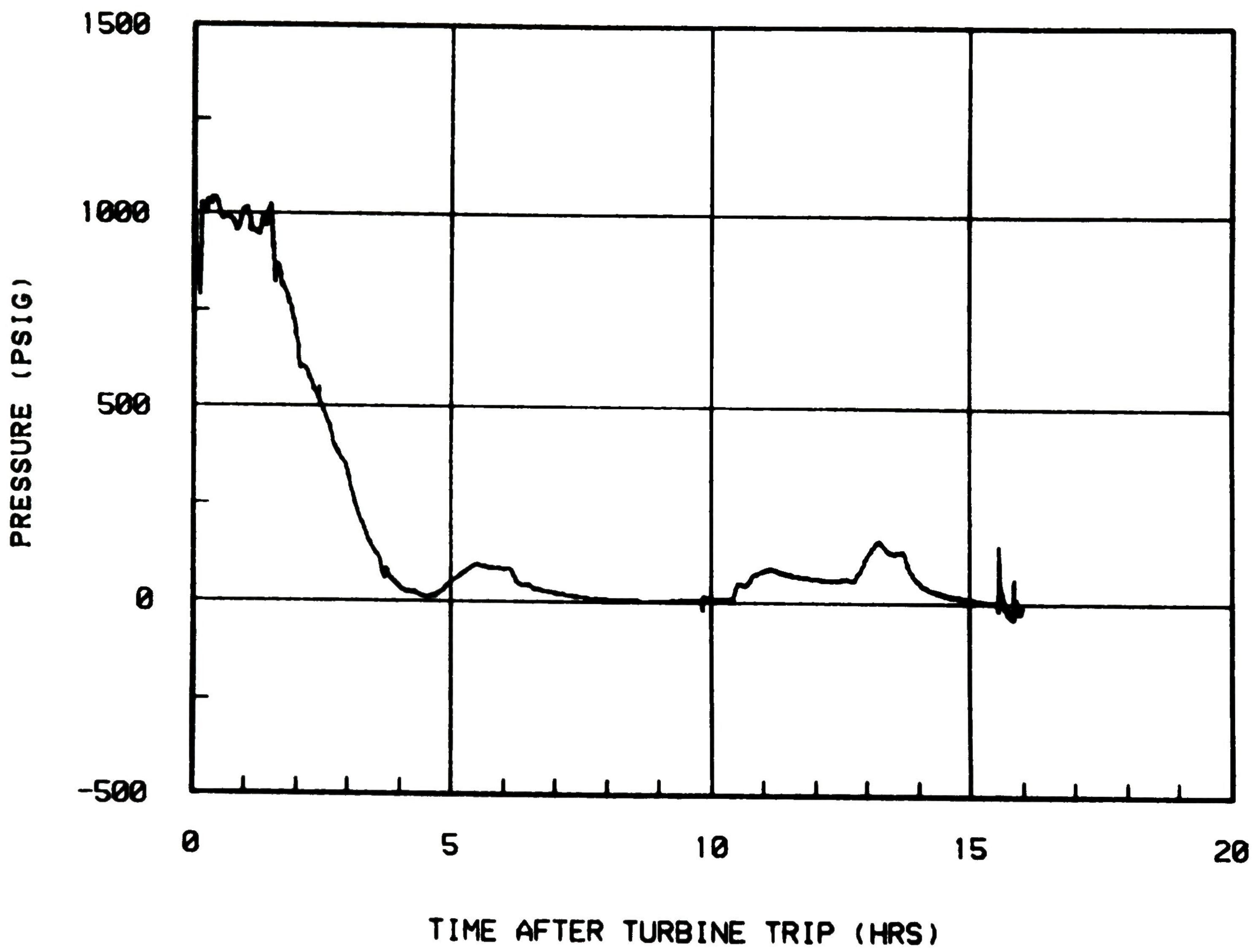

Figure 4. A-Loop Steam Generator Secondary Pressure -- Long Term.

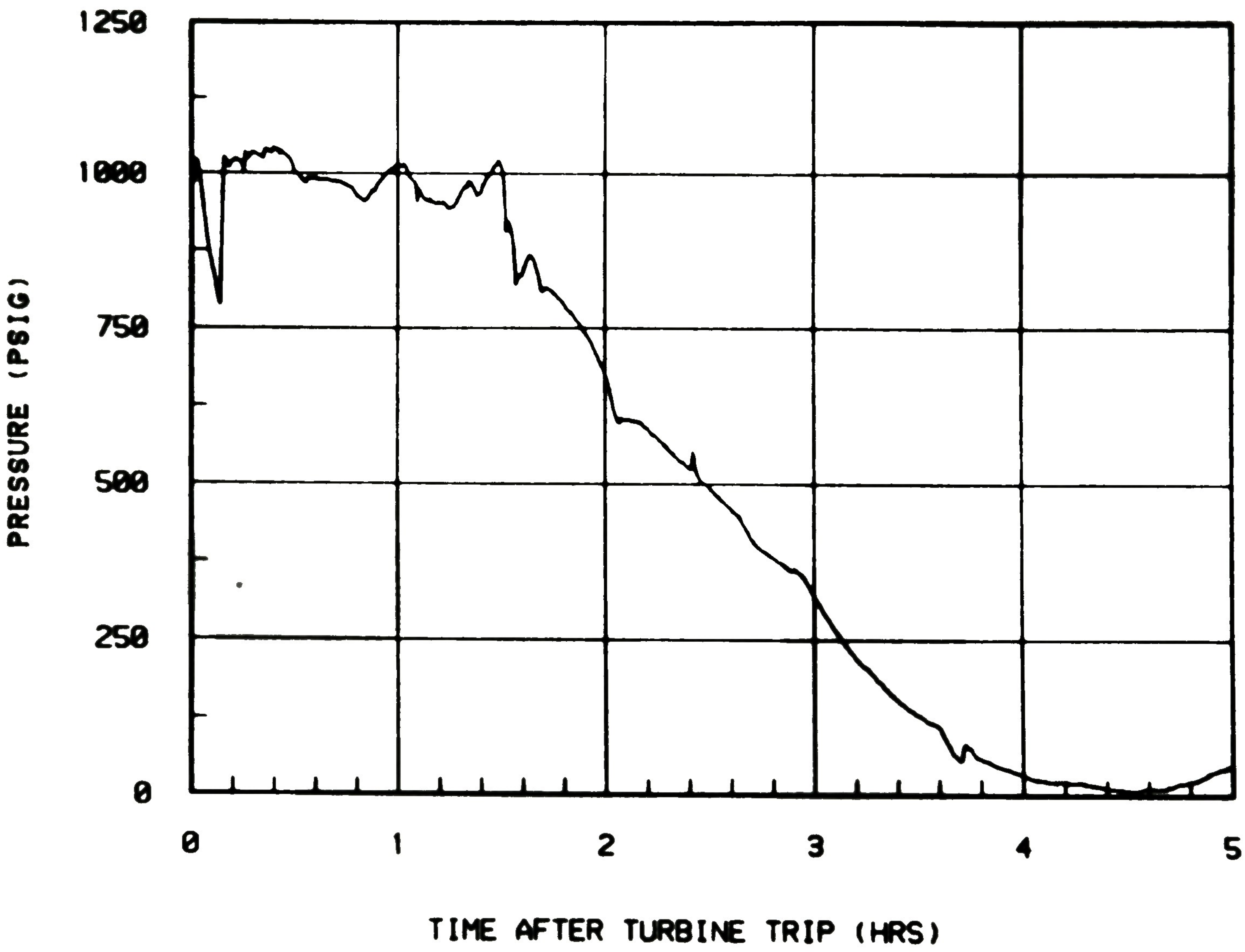

Figure 5. A-Loop Steam Generator Secondary Pressure -- Short Term.

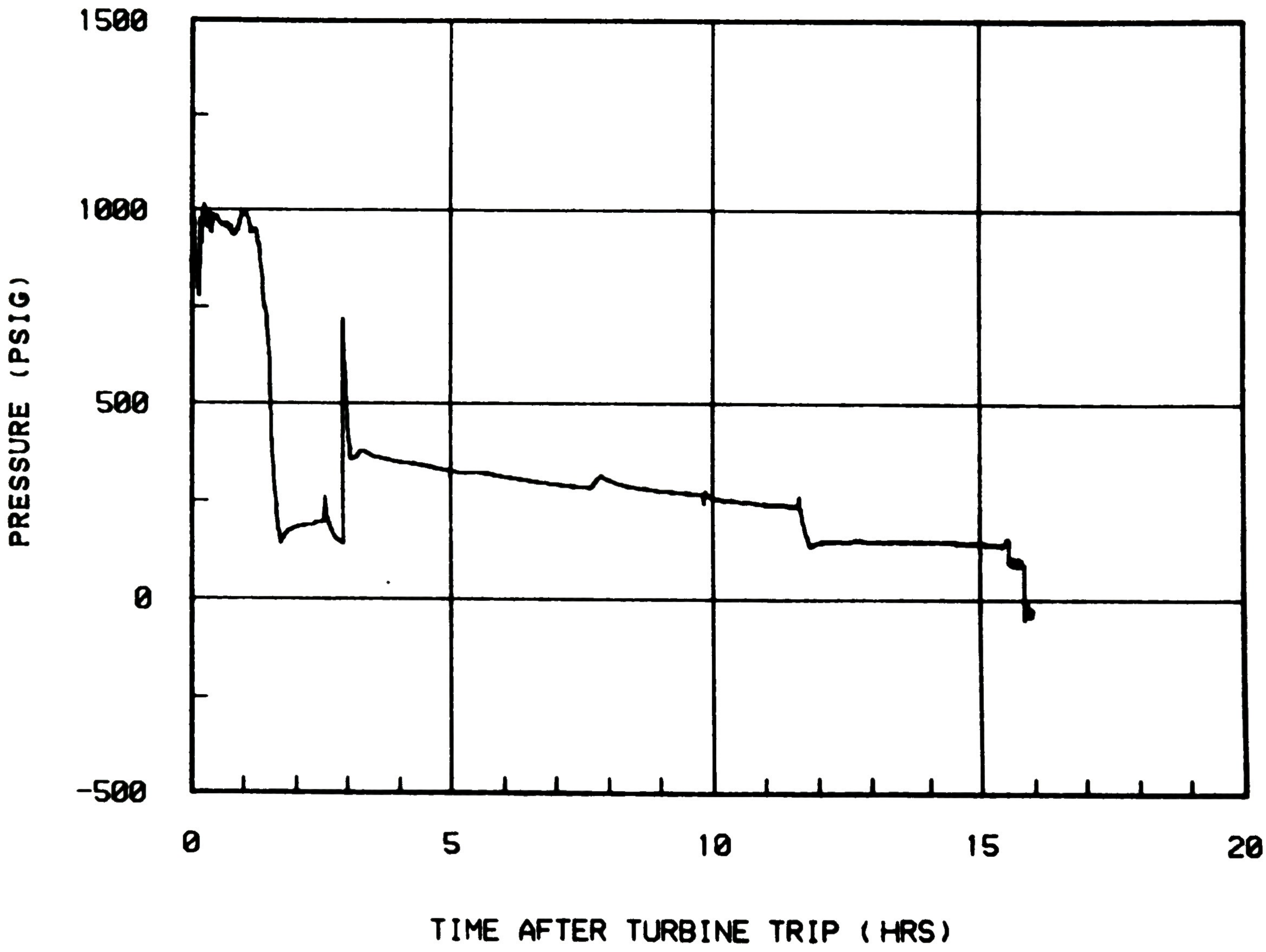

Figure 6. B-Loop Steam Generator Secondary Pressure -- Long Term.

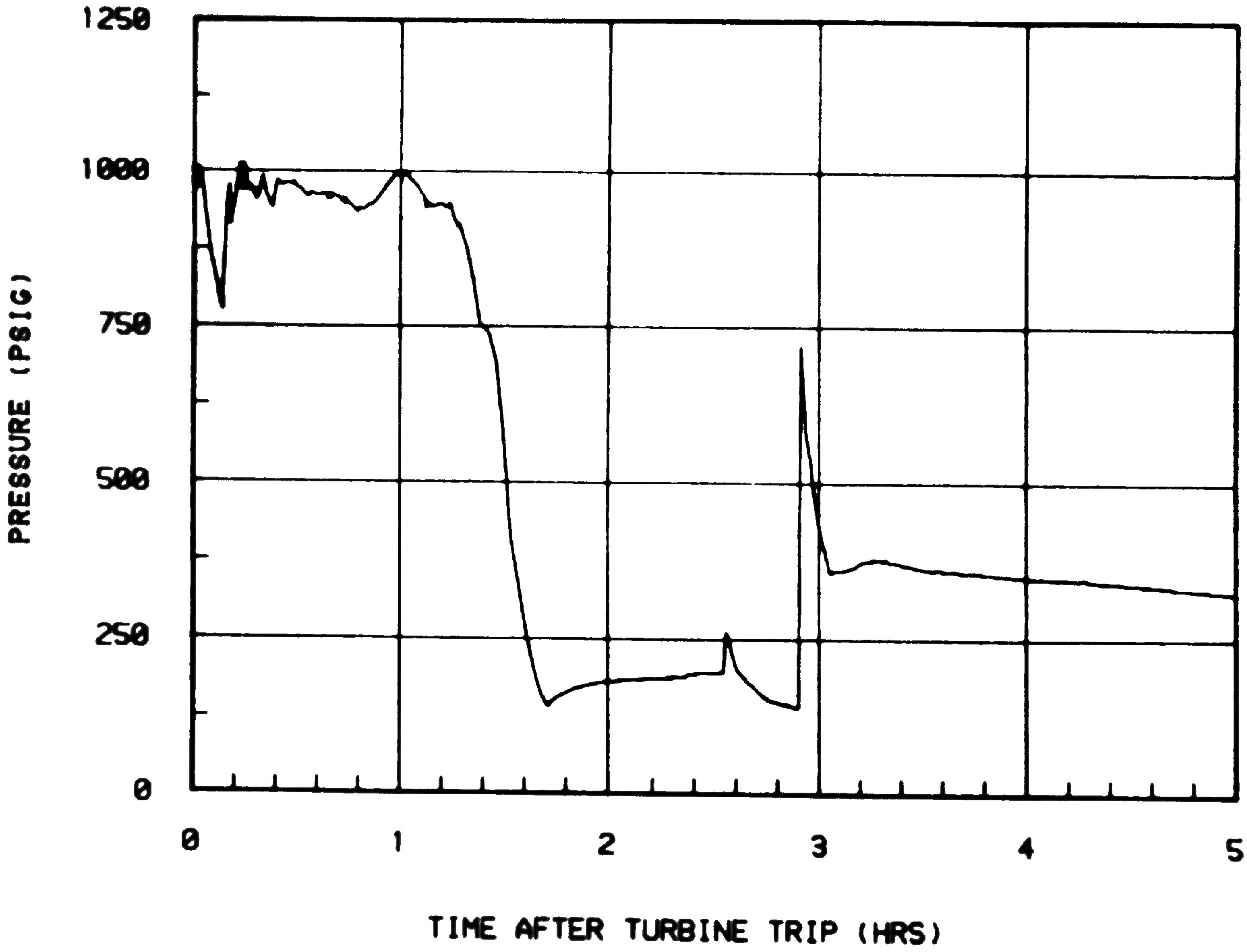

Figure 7. B-Loop Steam Generator Secondary Pressure -- Short Term.

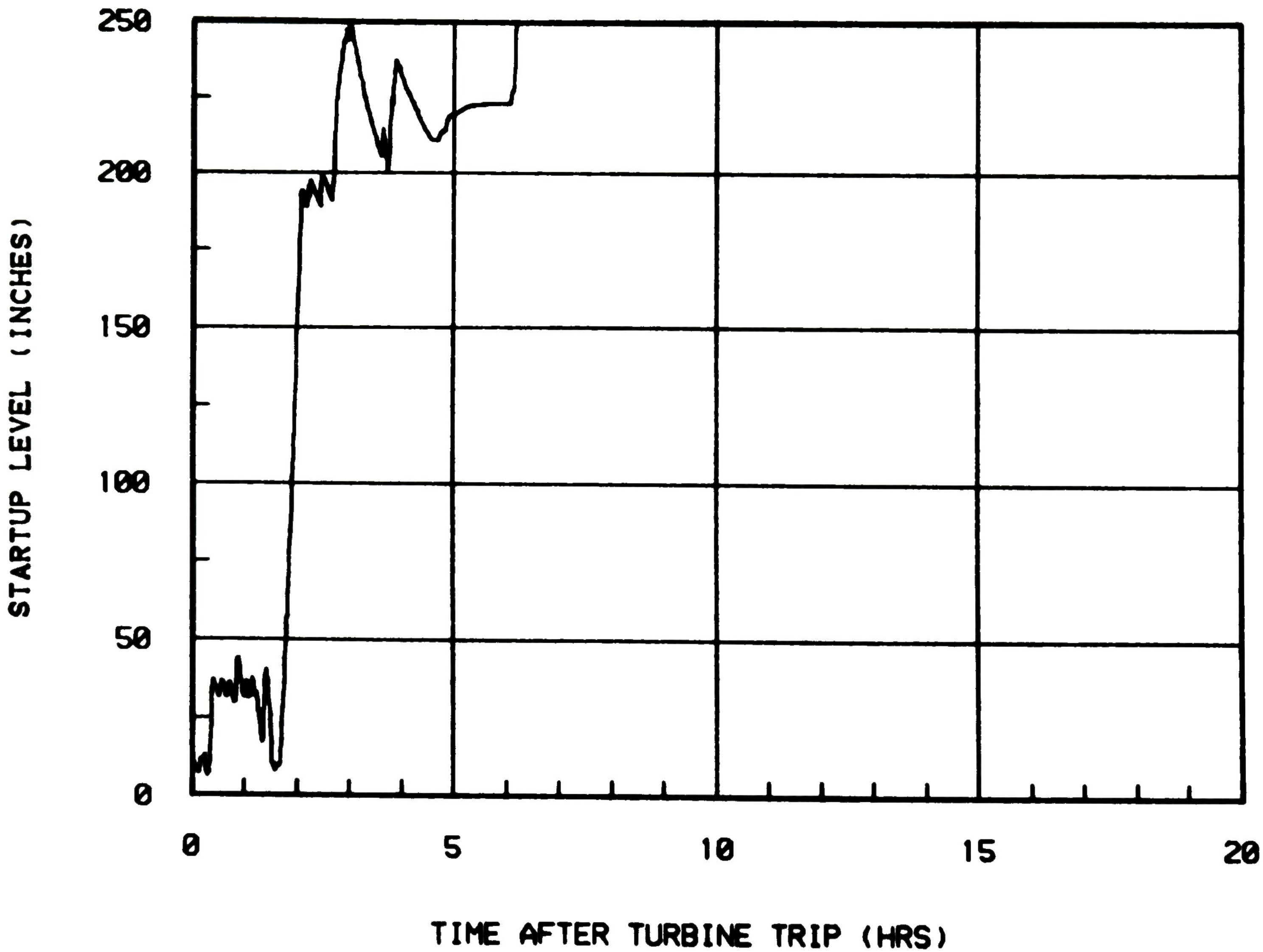

Figure 8. A-Loop Steam Generator Startup Level -- Long Term.

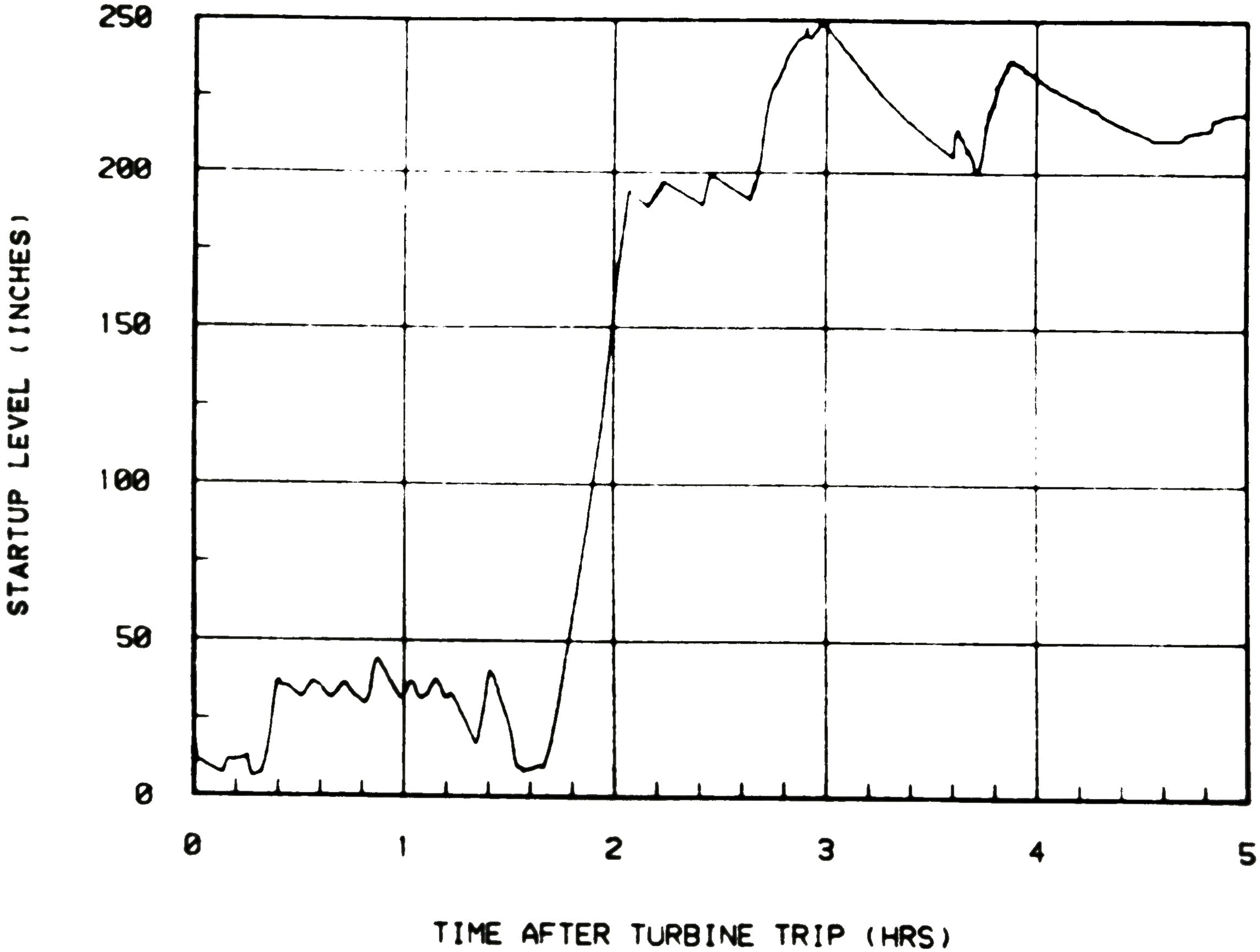

Figure 9. A-Loop Steam Generator Startup Level -- Short Term.

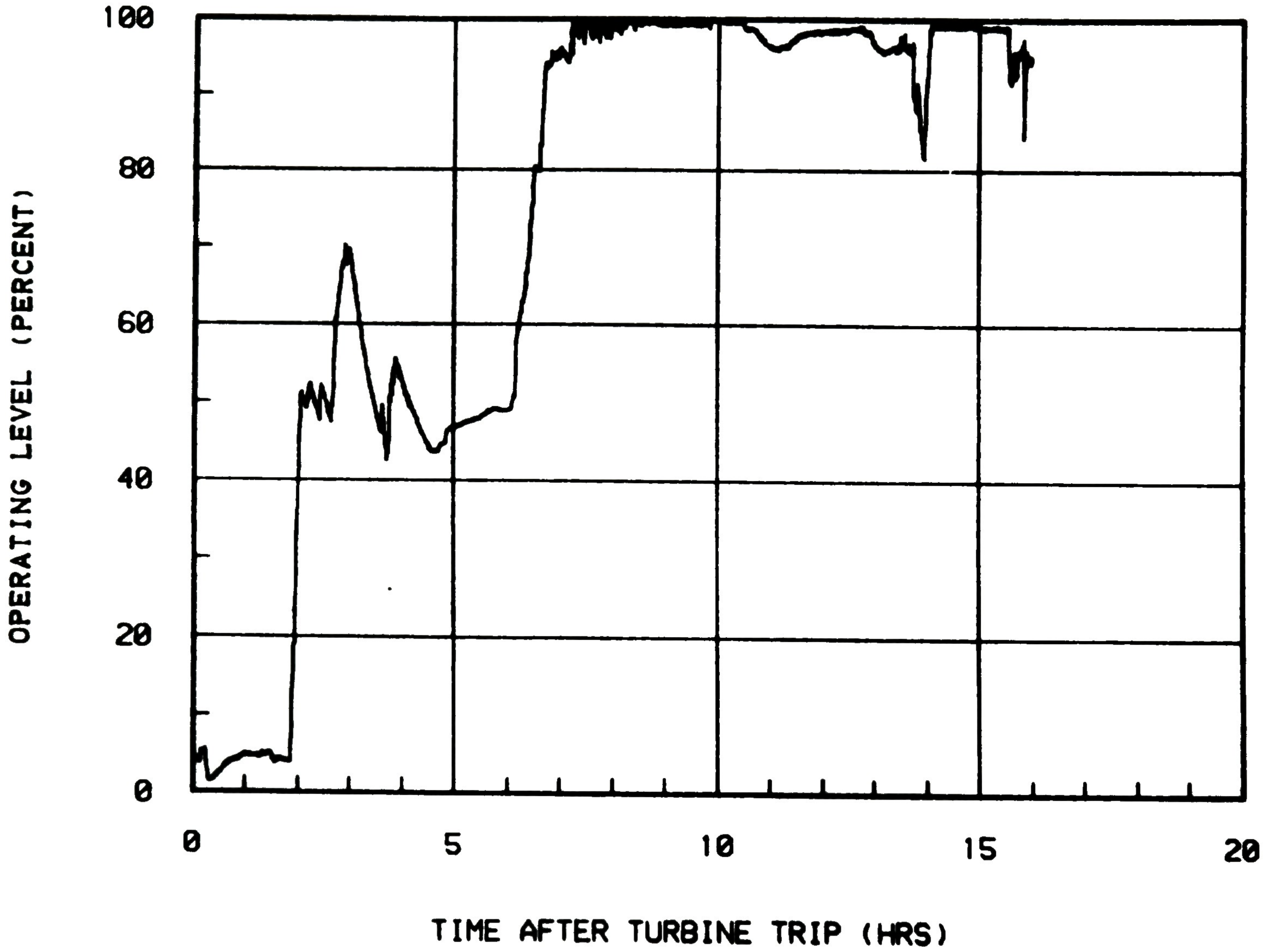

Figure 10. A-Loop Steam Generator Operating Level -- Long Term.

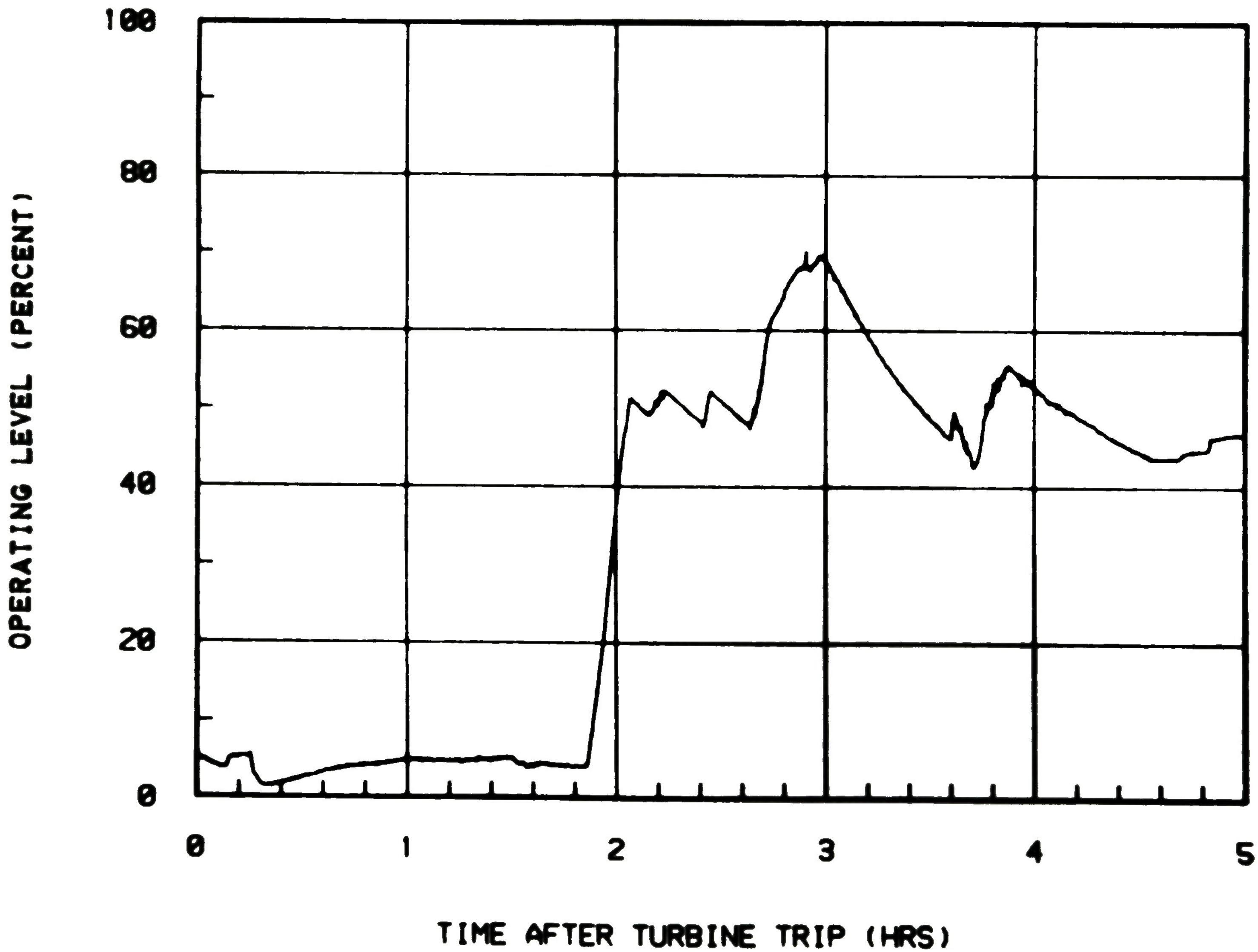

Figure 11. A-Loop Steam Generator Operating Level -- Short Term.

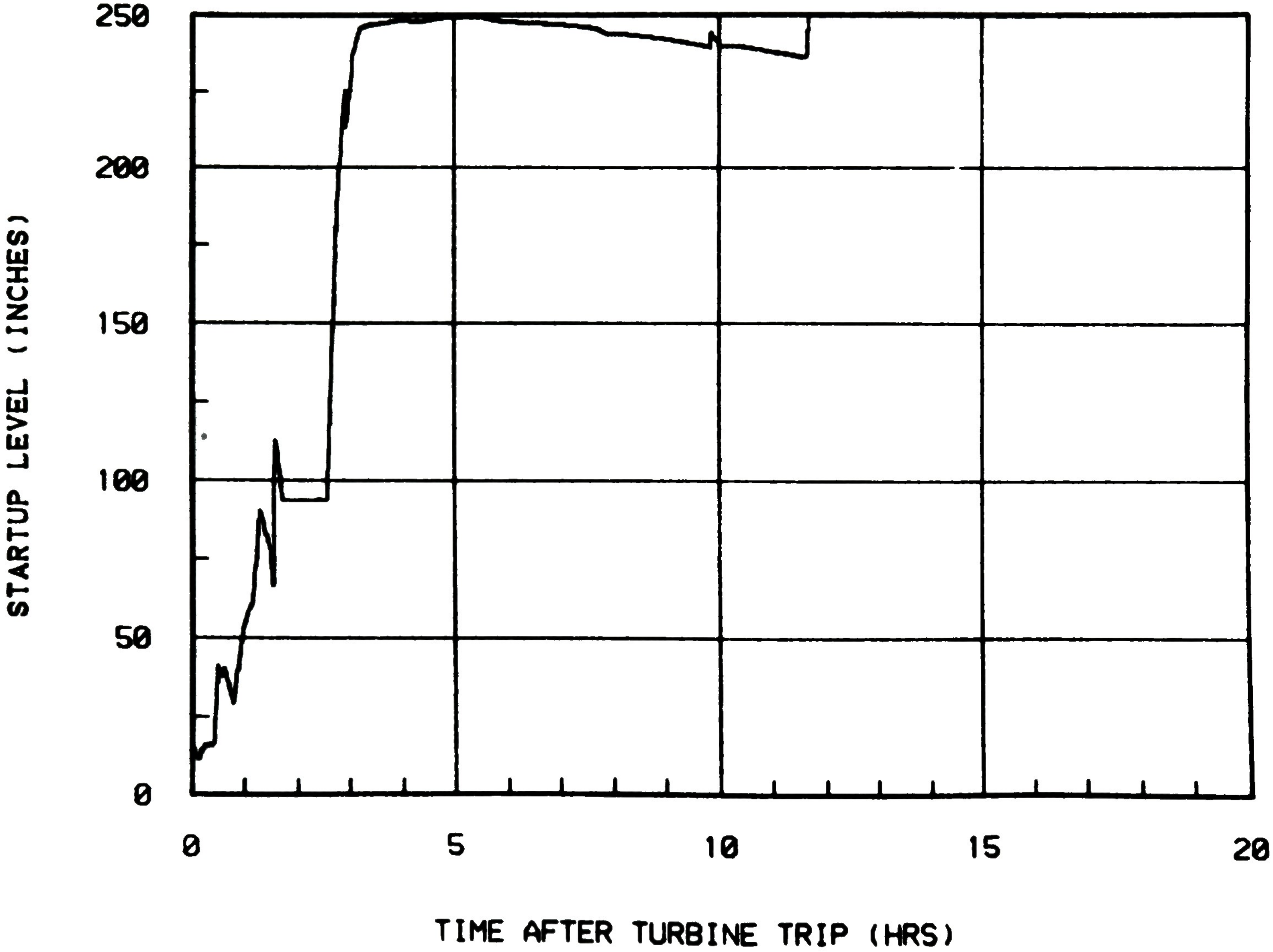

Figure 12. B-Loop Steam Generator Startup Level -- Long Term.

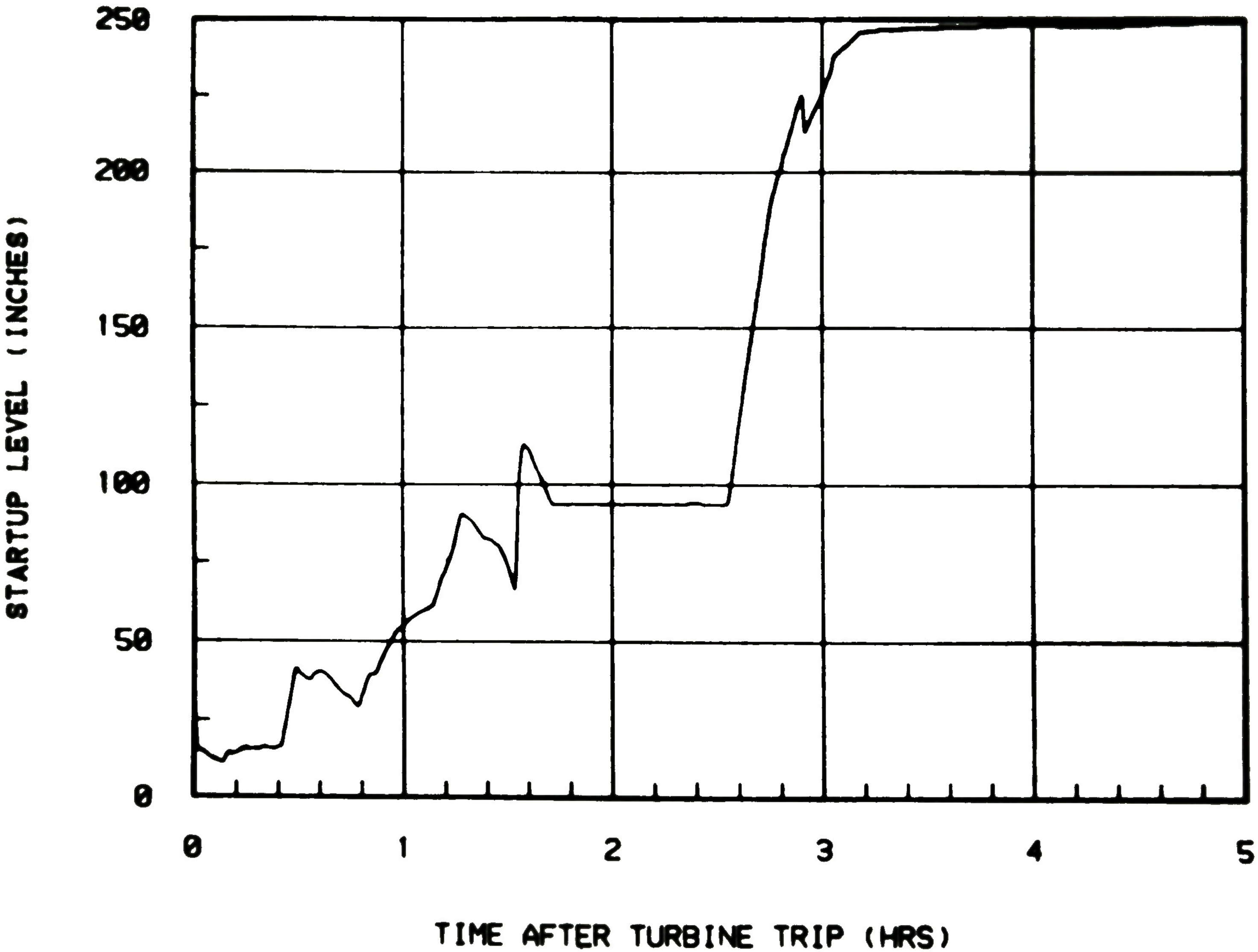

Figure 13. B-Loop Steam Generator Startup Level -- Short Term.

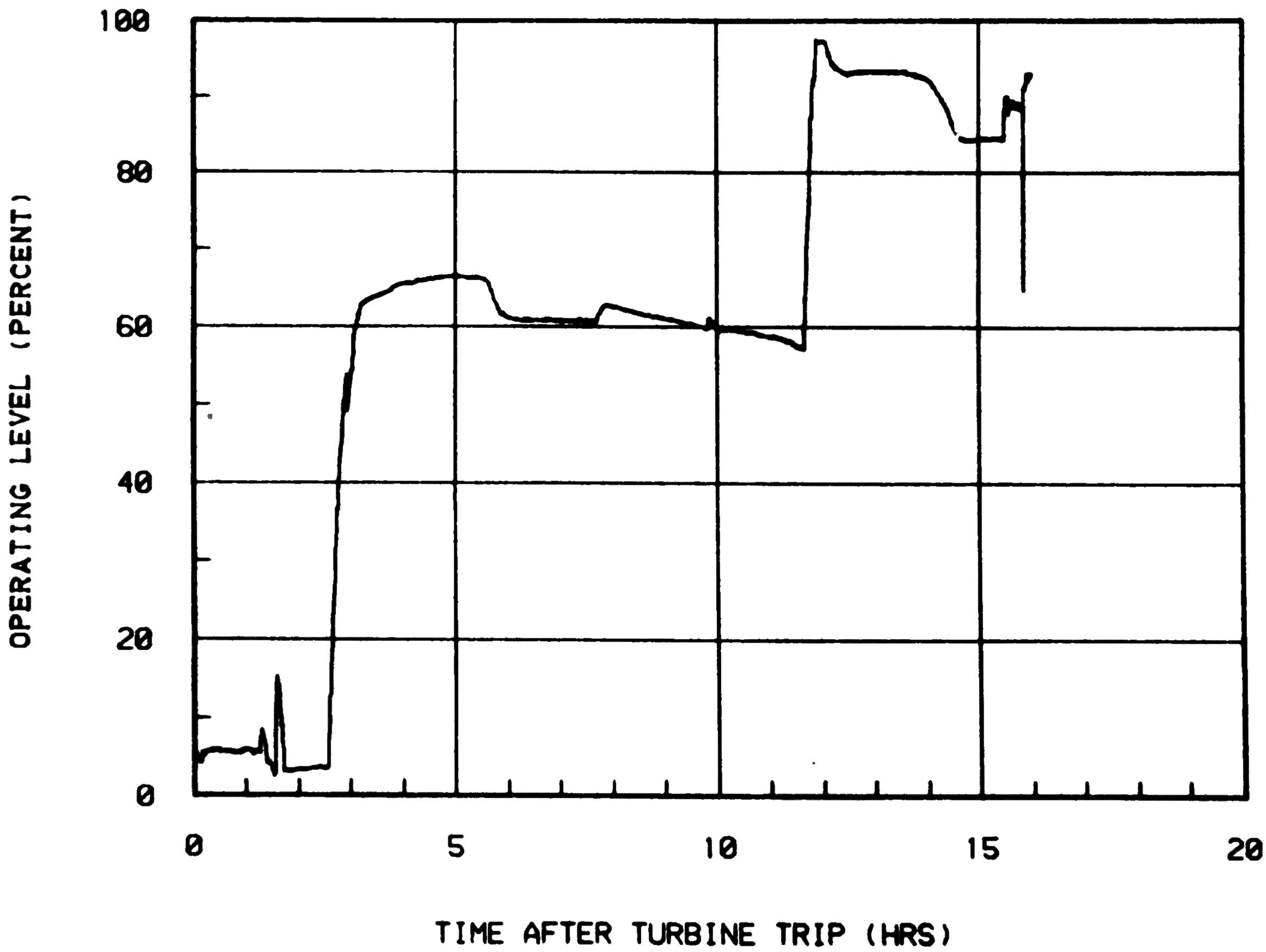

Figure 14. B-Loop Steam Generator Operating Level -- Long Term.

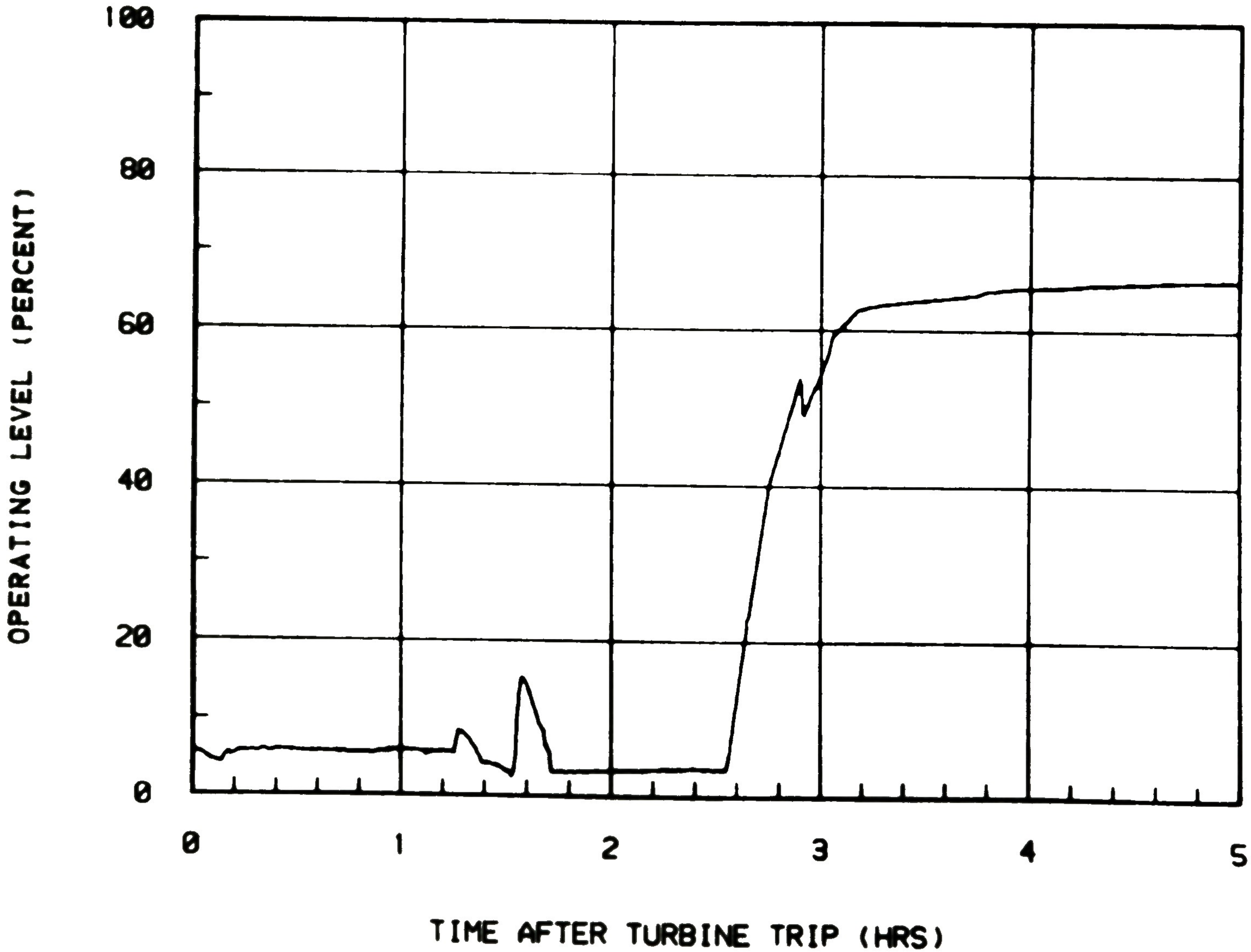

Figure 15. B-Loop Steam Generator Operating Level -- Short Term.

6. SUMMARY

This document has compiled a best estimate of initial and boundary conditions during the TMI-2 accident, with emphasis on those parameters needed to represent the progression of the accident in the primary coolant system. Not all important parameters are known precisely and many have very high uncertainties associated with them. Further research will improve this situation.

Analysis of the TMI-2 accident will be difficult, due to the uncertainties in the initial and boundary conditions and the extremely complex, coupled phenomena taking place. To realistically analyze the accident on a systems basis requires detailed representation of the plant and the use of highly developed mechanistic tools that couple the various physical phenomena occurring during the accident progression. In many areas, such as molten core interaction with lower core support structures, the mechanistic modeling is not yet in place and reliance must be placed on sound engineering analyses.

The boundary conditions described in Section 5 controlled the progression of the accident. The large uncertainties in these conditions can lead to calculate responses that are very much different, when the extremes of the uncertainty ranges are used. It is anticipated that analysis of the TMI-2 accident will require more than one calculation to bound the response of the system. As the uncertainties in the boundary conditions are decreased through research, the calculated response of the system will become more certain. As these advances are made, additional reports with the improved initial and boundary conditions will be issued.

7. REFERENCES

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4. K. H. Ardron, D. G. Cain, TMI-2 Accident Core Heat-up Analysis, Nuclear Safety Analysis Center report NSAC-24, January 1981.

5. EPRI PWR Safety and Relief Valve Test Program: Safety and Relief Valve Test Report, EPRI Special Report NP-2628-SR, December 1982.